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EXPERIMENTAL STUDY OF PHYSICAL CHARACTERISTICS OF WATER-MODERATED REACTORS FOR SMALL NUCLEAR POWER STATIONS

E.I.Inutin, M.N.Lantsov, I.G.Morozov,
A.I.Nevinitsa, E.A.Plaksin, G.I.Sidorov

Experimental study results for some lattices characteristics and presented in the paper. The lattices are consisted of tubular uranium(10-20% U^{235} enriched) fuel elements and light water as a moderator. The experimental study is a part of the research program which is underway in the Soviet Union on development of different type reactors for low power nuclear stations /1,2/.

A special paper section considers systematization of data on critical masses of uranium-water lattices. The semiempirical critical equations obtained from such systematization are used to analyse experimental data on the temperature reactivity coefficients and excess reactivity.

I. CRITICAL EXPERIMENTS WITH UNDISTURBED COLD LATTICES

Experimental plant. The majority of the experiments the results of which are given below were performed in a special critical facility. The critical facility permits to perform experiments in cores with a diameter of 600 mm and a height of 800 mm for temperature ranging 15°C to 230°C.

The main plant element is a stainless steel high pressure vessel with a diameter of 700 mm and a height of 1800 mm designed to operate at pressure of 40 atm. At the bottom of the vessel there is installed a band electrical heater of 75 kWh capacity. The upper portion of the vessel volume serves as a volume compensator in the experiments with heating of the lattices.

The experiments were made with four types of tubular fuel elements, basic parameters of which are presented in Table I.

Critical masses. The studied lattices assembled in the vessel of the experimental plant were regular, with no local perturbations and were of form close to a right circular cylinder. The vessel dimensions were such that in the experiments the lattices were surrounded by an infinite three-sided water reflector. The upper reflector was formed by an array of the fuel element dry ends. Criticality as rule was achieved through vessel filling by water by means of a dosing system. Simultaneously with the critical mass measurement the value of $\frac{d\rho}{dh}$ was measured for each critical height (ρ-the reactivity, h-the core height corresponding to a given critical radius $-r_{cr}$). The measurements enabled values of the upper and bottom water reflector summary savings (δ) and $M^2/K\infty$ to be determined in the one group diffusion approximation. The experimental data on the critical charges, $d\rho/dh$ and δ are given in Fig. I and 2.

Disadvantage factor for thermal neutrons. A disadvantage factor(the average thermal neutron flux in a moderator to the average flux in a fuel ratio) in the diffusion approximation can be written as $d=d_1 + d_2$ α is a term due to fuel self-shielding. d_2 is a term determined by the neutron flux non-uniformity in a moderator.

For fuel elements similar to used $d_1 \approx 1$, d_2 is determined by a fuel element construction and a lattice pitch, and by the thermal neutron diffusion length in water. In complex lattices the calculation of d_2 is not sufficiently reliable, and therefore it is required an experimental verification of calculational methods.

The disadvantage factor was measured in the central cell of lattices studied from activation of a copper foil with a thickness δ =80 μ accompanied by the induced activity measurement with a system of two end-window β -counters. The results of the disadvantage factor measurements at

20°C for the fuel elements of three types are presented in Table II, as are the calculational data.

There is a very small disadvantage factor in the regular lattices composed of M-10 and M-20 type fuel elements. However, it is assumed the fuel elements of these types to use in concrete reactors as fuel assemblies. Therefore water gap effect on the disadvantage factor have been studied for one of the lattices composed of the type M-10 fuel elements (ρ_H / ρ_S =275). The lattice was assembled with a water gap in the centre of it which had a width from 8 to 30 mm. The disadvantage factor was determined in a plane geometry. An area of regular lattice in which the "water gap" effect was still observed was taken as the fuel assembly "size". The dependence of the disadvantage factor upon the water gap width is given in Fig.3.

Comparison of experiments with a theory. The critical charges and thermal neutron disadvantage factor values obtained from experiments were compared with the values calculated. The calculations were made on the basis of a spherical harmonic method, an algorithm of which was described in reference /3/. The immediate calculations were conducted with using of proper programs on a digital computer as the 18-group calculation in a P_1 -approximation. Kinetic corrections were obtained from a P_3- approximation. As it should be expected, the experimental and calculated values of k_{eff} are in good agreement for systems with the ratio $\gamma cr / hcr$ $\sim 0,5$. Observed discrepancy of the calculations and the experiments at the values $\gamma cr / hcr$ far from 0,5 indicates the inadequacy of the accepted calculational model for describing of the space-energy neutron distribution. Such improvement as an account for heterogeneous effects in the resonance area and the multigroup thermalization lead only to a slight improvement of the results. An application of two-dimensional calcula-

tion with using of approximations of more high order makes it permitable a better agreement of the calculational and the experimental results in a wider range of the values Y_{cr}/h_{cr}

II. TEMPERATURE REACTIVITY COEFFICIENTS

In this paper section results are presented of temperature effects study for static conditions (for uniform heating of a core and reflectors). The measurements were conducted in critical assemblies the cores of which were assembled from the M-10, M-20 and K-17 type fuel elements. The critical assemblies composed of the M-10 and M-20 type fuel elements were in their characteristics near physically homogeneous systems, yet the critical assemblies composed of the K-17 type fuel elements were of pronounced heterogeneous structure. The M-10 and M-20 type fuel elements permitted heating to 100°C only. The K-17 elements permitted heating to 300°C.

Disadvantage factor dependence upon temperature. In water-moderated reactors with pronounced heterogeneous core structure temperature changes in the disadvantage factor give the main contribution to the quantity $d K_\infty / dt$ and, hence, to the temperature reactivity coefficient. For several lattices assembled from the K-17 type fuel elements the disadvantage factor was studied as a function of the core temperature and the ratio $\rho_H / \rho_S \, (V_{H_2O} : V_U)$

The results of the measurements of the disadvantage factor temperature dependence for three lattices in the range of 20°C -200°C are presented in Fig. 4. The continuous lines are the calculational results obtained for the variables by the method described in section I of the paper.

As it is evident from the curves the calculational data are in a qualitative agreement with the experimental results, however, a quantitative agreement is inadequate. A dashed line is experimental results processed in a P_1 - approximation

for a cylindrical block using the experimental temperature dependence of the thermal neutron diffusion length in water /4/. Such an approximation is in better agreement with experiment and therefore the results were used to calculate dk_∞/dt for the lattices studied.

Temperature effect. In several undisturbed lattices, which differed one from another for the most part by the degree of their heterogeneity, the effect of temperature on reactivity of system was studied. The critical assemblies were heated by an external heat source and thereactivity change was measured as a function of temperature.

In the physically homogeneous cores (the case of the M-10 and M-20 type fuel elements), the temperature reactivity coefficient for which is negative in the whole of temperature range, reactivity measurement in heating was performed by a subcritical reactor method. Point of the method is as follows:

The reactor kinetic equation solution with the external neutron source (S) can be expressed in terms of external detector counting rate ($N=n\xi$, where N-the external detector counting rate, ξ -the counter effectiveness, n-the neutron density in a reactor). The solution for a subcritical reactor with the initial conditions.

$$\frac{dn}{d\tau}=0 \quad and \quad \frac{dC_i}{d\tau}=0$$

is of the form

$$\Delta K_{eff} = \frac{S L_f \ell \xi}{N} \qquad (1)$$

The quantity $A_n = L_f \ell$ is determined from the critical reactor experiments. For a critical reactor with the initial conditions.

$$n/_{\tau=0} = n_o \qquad C_i\big|_{\tau=0} = C_{io}$$

$$\frac{dn}{d\tau}\Big/_{\tau=0} = S \qquad \frac{dC_i}{d\tau}\Big|_{\tau=0} = 0 \qquad (2)$$

359

$$N = \frac{S\pi L_f \ell \xi}{\ell + \sum_{i=1}^{6} \beta_i/\lambda_i}$$

Neglecting by 1 in comparison with $\sum_{i=1}^{6} \beta_i/\lambda_i$ and introducing the notation $A_k = L_f \ell / \sum_{i=1}^{6} \beta_i/\lambda_i$ it can be found from (3) and (4)

$$A_n = A_k \sum_{i=1}^{6} \beta_i / \lambda_i$$

Thus the equation which relates the reactor subcriticality to the detector counting rate may be written conclusively as *

$$\Delta k = \frac{A_k \sum_{i=1}^{6} \frac{\beta_{i\,ef}}{\lambda_i}}{N} S\xi \qquad (3)$$

A neutron source is placed near the critical reactor and a curve of counting rate increasing is obtained as a function of time. The product $A_k S\xi$ equals the tangent of the dip angle for the straight line corresponding to reactor power increasing.

The present method of measurements requires highly accurate organization of experiments. In particular a number of detectors uniformly located over critical assembly periphery are required, and a background must be taken into account especially in measurements in case of deep subcriticality.

In the heterogeneous assemblies which used the K-17 type fuel elements at first the reactivity was increasing in heating and only after achieving of a certain temperature the reactivity was decreasing. Therefore in heating of these lattices another method of measurements was used. In the initial ($t=20^{\circ}C$) the critical assembly was brought exactly to criticality. After this the heater was set off and the system was heating by $5\text{-}10^{\circ}C$, the temperature was stabilizing, and the reactor period was measured. Then the system was brought to criticality by inserting in the core of absorbers which featured little or no effectiveness changes with temperature, and the system was heating by $\Delta t = 5 + 10^{\circ}C$ again. In the experiments B_4C rods placed inside tubular fuel elements and

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* A similar method of the negative reactivity measurement to apply for determination of the absorbing rod effectiveness was proposed in reference /5/.

stainless steel tubes enveloping all fuel elements were used.This way the measurements were made on the ascending branch of the curve $\rho = f(t^{\circ}C)$.Simultaneously with temperature coefficient measurement calibrating of the absorbers was going on.The measurements on the descending branch of the curve were performed by means of the calibrated absorbers or from reactor period measurement as the system was getting cold.

In Fig.5 the results of the experiments with heating of the critical assemblies are presented.

To illustrate a leakage contribution to temperature effect the curves of heating are shown in Fig.5 a and 5b for the cores assembled from the K-17 type fuel elements and differing one from another with a radius .The dashed lines are results of temperature curves calculations.A discussion of the calculational and experimental results is given in the next section of the paper.

III.SYSTEMATIZATION OF EXPERIMENTAL WATER MODERATED CRITICAL ASSEMBLIES

Derivation of empirical critical equations. An application of the low group approximations to small water moderated reactors ($k_{\infty} > 1,15$; $B^2 > 5.10^{-3}$ cm^{-2}) meets with a number of difficulties:

a) To determine the moderation kernel fourie form it is required to account not only for the second moments but moments of high order an experimental measurement accuracy of which is inadequate.

b) The extrapolation lengths of small reactors are comparable to core dimensions while the methods of their experimental determination have large errors.

From above there was proposed a method by Lantsov M.N., one of the authors of the paper, to obtain semiempirical critical equa-

tions through systematization of experimental water moderated critical assemblies as follows:

1) As a characteristic of core multiplicating properties a calculational value of (k_∞ -1) was used.

2) As a geometrical factor, which determines the neutron leakage dependence upon core dimensions, the core surface to core volume ratio (s/v) was chosen.

3) Through systematization the dependence $\frac{s}{v} = f(k_\infty - 1)$ was found with k_∞ determined as the secondary neutron number per an absorption. For all critical assemblies a common scheme of calculations was used.

The systematization with respect to the parameter s/v was performed on the base of the present paper data and experimental results published before on uranium-water homogeneous and heterogeneous critical assemblies of various geometry with $k_\infty \geq 1{,}15$ /6-9/. Critical equations were obtained for various sets of critical assemblies.

The systematization of heterogeneous critical assemblies includes only ones for which experimental data could be obtained on disadvantage factors and a fraction of U^{238} fissions.

Cylindrical critical assemblies were preliminarily analysed with respect to carrying-out of the condition s/v = const as the core dimensions ratio (r_{cr}/h_{cr}) is changed. Critical assemblies which satisfied the condition within an error of no more than ± 1% were included in the systematization. It is turned out that the critical assemblies with $0{,}4 < r/h < 1$ satisfied the condition. The heterogeneous assembly radius was determined from the summary area of core cells.

In Fig.6 the results of the systematization of critical water moderated assemblies are presented graphically. The analytical results of the systematization reduce to the following empirical critical equations.

I) For spherical homogeneous critical assemblies with

359

no reflector

$$\frac{S}{V} = 3/r = 0,048 + 0,166(k_\infty - 1) \qquad (4)$$

2)For spherical water reflected homogeneous critical assemblies

$$\frac{S}{V} = 0,038 + 0,239\,(k_\infty - 1) \qquad (5)$$

with $0,23 < (k_\infty - 1) < 0,92$

3)For cylindrical homogeneous critical assemblies with no reflector

$$\frac{S}{V} = 2/r + 2/h = 0,053 + 0,185\,(k_\infty - 1) \qquad (6)$$

4)For cylindrical homogeneous critical assemblies with a full water reflector

$$\frac{S}{V} = 0,046 + 0,26\,(k_\infty - 1) \qquad (7)$$

5)For cylindrical homogeneous critical assemblies with a three-sided reflector

$$\frac{S}{V} = 0,04 + 0,246\,(k_\infty - 1) \qquad (8)$$

with $0,15 < (k_\infty - 1) < 0,93$

6)For cylindrical heterogeneous critical assemblies with $\frac{V_{H_2O}}{V_u}$ and a three-sided water reflector

$$\frac{S}{V} = 0,026 + 0,243\,(k_\infty - 1) \qquad (9)$$

with $0,4 < (k_\infty - 1) < 0,61$

Thus in the considered range of $(k_\infty - 1)$ the all critical equations is linear and may be written in the following general form:

$$\frac{S}{V} = C + (C_2 + C_3)\,(k_\infty - 1) \qquad (10)$$

where

C_1- a neutron core leakage characteristic, is a function of C_3 and core geometry;

C_2- an independent neutron core leakage:

C_3-a generalized reflector characteristic independent upon both core properties and core geometry.

For the large values of $(k_\infty - 1)$ the factor $\frac{S}{V}$ is determined for the most part by the coefficients C_2 and C_3, as $C_1 \ll C_2$.

359

The C_3 universality makes convenient a comparison of various reflectors. In case of such universality for other reflector materials

$$\frac{C_3^*}{C_{3H_2O}} = \frac{\delta_x (r - \delta_{H_2O})}{\delta_{H_2O}} \qquad (II)$$

where r -the critical size of an unreflected critical assembly; δ_x , δ_{H_2O} -the differences in critical sizes of a critical assembly unreflected and with a given reflector.

To obtain the calculational dependence $\frac{S}{V} = f(k_\infty - 1)$ the equation

$$k_\infty - 1 = \frac{1 - p + \varkappa^2 L^2}{p} \qquad (I2)$$

was solved with a given $\frac{S}{V}$ and a variable value of λ_0 -the exrapolation length. To calculate the value of the non-leakage probability (p) at slowing-down process the experimental slowing-down kernels in water /I0-13/ were used, the Fouri form of the slowing -down kernel being determined from experimental moments.

The neutron non-leakage probability calculated from the data of reference /I0/ is as

$$p = 0{,}443 + \frac{0{,}063}{B} \arctan 8{,}78\,B - 16{,}5\,B^2 + 3{,}56 \cdot 10^2 B^4 - 8{,}5 \cdot 10^3 B^6 + 1{,}9 \cdot 10^5 B^8 \qquad (I3)$$

if the data of references /II-I3/ afe used

$$p = 0{,}50 + 0{,}90 \left[\frac{0{,}063}{B} \arctan 8{,}78\,B - 16{,}5\,B^2 + 3{,}56 \cdot 10^2 B^4 - 8{,}5 \cdot 10^3 B^6 + 1{,}9 \cdot 10^5 B^8\right] \qquad (I4)$$

In Fig.7 there are given the results of comparison of the sistematization data with the calculational predictions of the neutron non-leakage probability (p) through the use of the experimental slowing-down kernel Fourie-form in water. The contributions of thermal neutrons and slowing-down neutrons to leakage are separated.

From the figure it is seen that the experimental critical assemblies systematization results and the calculation of p performed using relations (I3) and (I4) are in good qualitative agreement. In the considered range of ($k_\infty - 1$) the calculation also predicts the linearity of the dependence

359

$\frac{s}{v} = f(k_\infty - 1)$ .In the best way the calculational and experimental dependences $\frac{s}{v} = f(k_\infty - 1)$ coincide if the slowing-down kernel described in reference /IO/ and τ_0 =const=3 cm are used.

Application of the empirical critical equations. The empirical critical equations obtained from the systematization for heterogeneous critical assemblies were verified by comparing with experimental data on reactivity margins for critical assemblies of a volume well over critical.

Such a reactivity margin may be determined experimentally by graphical integrating of the experimental curve $dp/dh = f(h_{cr})$ where $h_{cr.}$ changes through the critical assembly radius vafiation.

Applied to such an experiment the empirical critical equations results in

$$\Delta k = \frac{1}{C_2 + C_3}\left(\frac{2}{r_{cr}^{min}} - \frac{2}{r}\right) \qquad (15)$$

where r_{cr}^{min} - the minimum value of the cylinder critical radius with the uranium elements length given; r-a current radius value.

This relation predicts the quantitative results for the reactivity margins of the critical assemblies fore-going in the paper with an accuracy of the correction to account for the delayed neutron efficiency .Here it is important to emphasize that the linearity of the dependence between ΔK and $\left(\frac{2}{r_{cr}^{min}} - \frac{2}{r}\right)$,which follows from the systematization results,in most cases fulfils better than in the limits of the experimental errors.This moment is illustrated in Fig.8, for one of the critical assemblies studied.

In interpreting of the reactivity temperature changes in critical water moderated assemblies through the empirical critical equations it was considered the following physical picture:

a) The core neutron leakage changes with temperature only due to the water density effect on the slowing -down kernel moments.

b) The "infinite" water reflector decreases the temperature increase of the core leakage as well as it decreases the total leakage .In this case ($\frac{S}{V}$ = const).

$$\frac{dK}{dt} = \frac{dK_\infty}{dt} + \frac{\frac{S}{V}C_2 + C_3 C_1}{[C_2 \gamma(t) + C_3]} \qquad (I6)$$

where $\gamma(t)$ -the water density.

Accordingly

$$\Delta K = \int_{t_o}^{t} dK_\infty - \frac{\left(\frac{S}{V}C_2 + C_3 C_1\right)[\gamma(t_o) - \gamma(t)]}{[C_2\gamma(t_o)+C_3][C_2\gamma(t)+C_3]} \qquad (I7)$$

In the paper the dependence $dK_\infty/dt = f(t°c)$ was found by differentiating the following relation for K_∞

$$K_\infty = \nu[\varphi\Theta_f^T + (1-\varphi)\Theta_f^P] \qquad (I8)$$

where ν -the number of the secondary neutrons per fission;

φ -the resonance-escape probability in all materials of the core Θ_f^T, Θ_f^P -fission neutron absorption fractions in the thermal group and in the epithermal group relatively.

In this case the disadvantage factor temperature dependence chosen was experimental and in the calculation of dK_∞/dt all effects were taken into account contributions of which were less than a contribution of the thermal neutron distribution flattening over a cell due to temperature no more than by I0 times.

In Fig.5 the results of the calculation of $\rho = f(t°c)$ using relations (I6) and (I7) along with the experimental dependences $\rho = f(t°c)$ which are with no correction for the delayed neutron efficiency are presented.From the figure it is seen that matching of the experimental and calculational curves $\rho = f(t°c)$ can be achieved by multiplying the

experimental data by approximately constant factor, which is of the same order of magnitude as the correction for the delayed neutron efficiency and equals I,2 ÷ I,3.

The critical experiments reported in the paper were performed as an experimental basis for the reactor design of the small transportable nuclear power unit. Obtained experimental information allowed to improve calculational methods for such reactors and to lay down a program of subsequent physical experiments. The performed systematization of the critical assemblies reported to date promoted to this in large part.

359

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359

FUEL ELEMENTS CHARACTERISTICS

Table I

		Units	Type of fuel elements			
			M-I0	M-20	k-I7	k-20
I.	Fuel		UO_2	UO_2	UO_2	UO_2
2.	Enrichment	%	I0	20	I7	20
3.	U^{235} amount per I cm of length	g	0,I92	0,264	3,48	3,I0
4.	Number of rings		one	one	two	three
5.	Ring size in a can	mm	I3,4x2,6	I3,4x2,6	4I,8x2,9 29,2x2,9	4I,8x2,6 3I,6x2,6 22,2x2,6
6.	Can material		Stainless steel	Stainless steel	Stainless steel	Stainless steel

Table II

DISADVANTAGE FACTOR FOR THERMAL NEUTRONS

Type of fuel elements	M-I0	M-20		K-I7	
ρ_H / ρ_s	275	280	I85	2I7	252
d exper.	I,05 ± 0,02	I,04 ± 0,02	I,87 ±0,08	I,98±0,09	I,92 ± 0,09
d calcul.	-	I,00	I,72	I,87	2,03

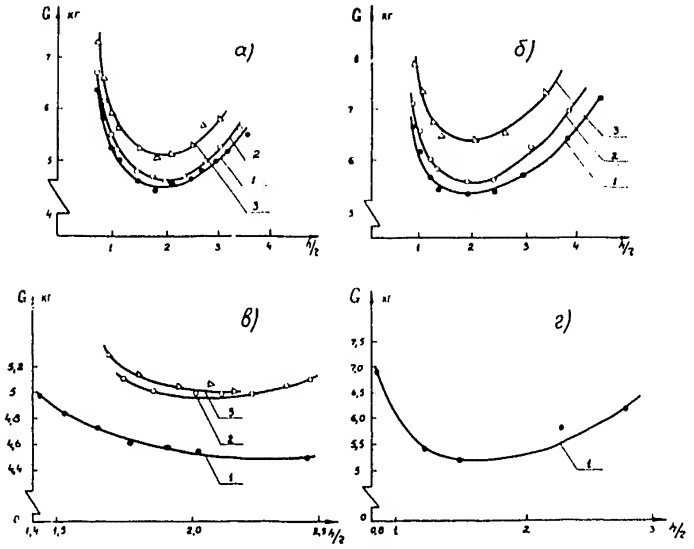

Fig.1. The dipendence of U^{235} critical charge on h/ρ_s and the ratio of core sizes(h/r).
a) K-17 fuel element 1-$\rho_н/\rho_s$=185; 2-$\rho_н/\rho_s$=217; 3-$\rho_н/\rho_s$=252.
b) K-20 fuel element 1-$\rho_н/\rho_s$=196; 2-$\rho_н/\rho_s$=232; 3-$\rho_н/\rho_s$=280.
c) M-10 fuel element 1-$\rho_н/\rho_s$=480; 2-$\rho_н/\rho_s$=275; 3-$\rho_н/\rho_s$=600.
d) M-20 fuel element 1-$\rho_н/\rho_s$=275.

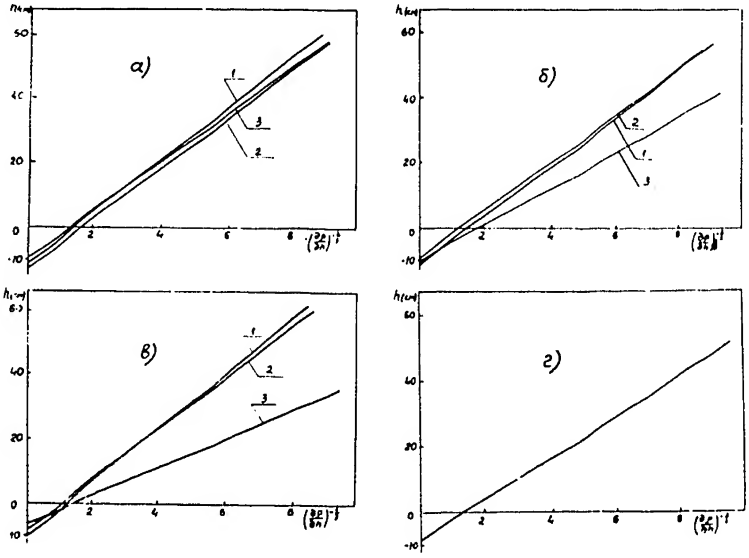

Fig.2. The $\partial\rho/\rho_э$ dependence on core height.
a) K-17 fuel element 1-$\rho_н/\rho_s$=185; 2-$\rho_н/\rho_s$=217; 3-$\rho_н/\rho_s$=252
b) K-20 fuel element 1-$\rho_н/\rho_s$=196; 2-$\rho_н/\rho_s$=232; 3-$\rho_н/\rho_s$=280
c) M-10 fuel element 1-$\rho_н/\rho_s$=480; 2-$\rho_н/\rho_s$=275; 3-$\rho_н/\rho_s$=600
d) M-20 fuel element 1-$\rho_н/\rho_s$=275.

359

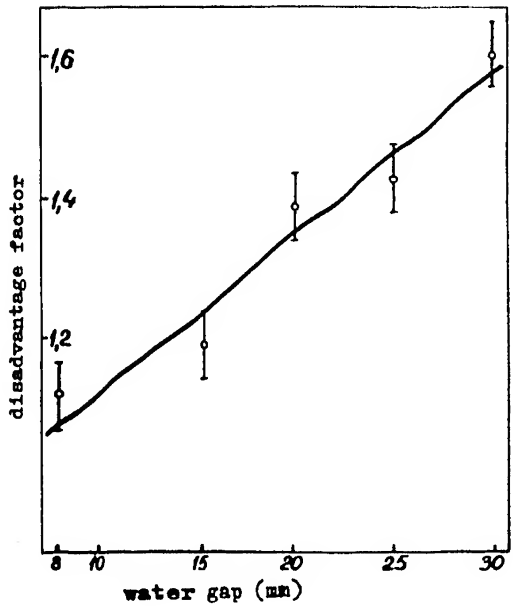

Fig.3. The dependence of disadvantages factor on the width of water gap.

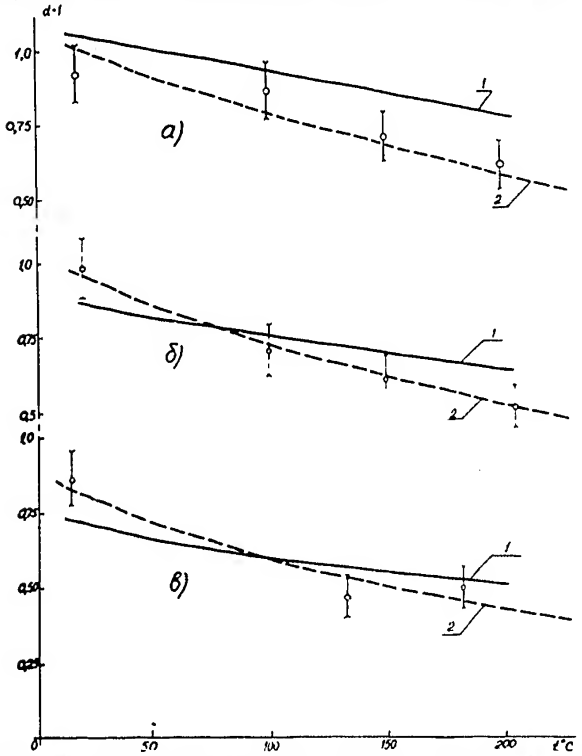

Fig.4. The temperature dependence of the disadvantage factor.
O-experiment; - calculation by the method described in section 1;
- - - - experimental results approximation.
a) $\rho_H/\rho_S = 2.52$; b) $\rho_W/\rho_S = 2.17$; c) $\rho_H/\rho_S = 1.85$.

359

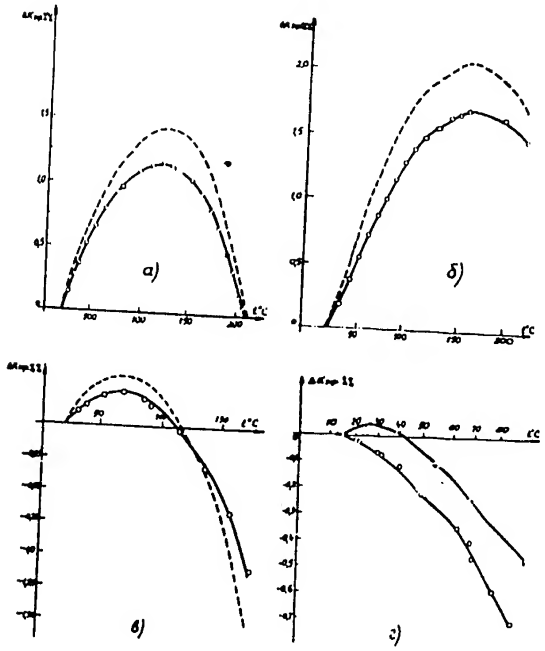

Fig.5. The temperature dependence of reactivity.
- experiment; - - - calculation using relation (17);
a) K-17 fuel element $\rho_n/\rho_s = 2.52$; $S/V = 0{,}143$ cm; b) K-17 fuel element $\rho_n/\rho_s = 2.40$; $S/V = 0.044$ cm;
c) K-17 fuel element $\rho_n/\rho_s = 1.85$; $S/V = 0{,}162$ cm; d) 0-M-10 fuel element; x-M-20 fuel element.

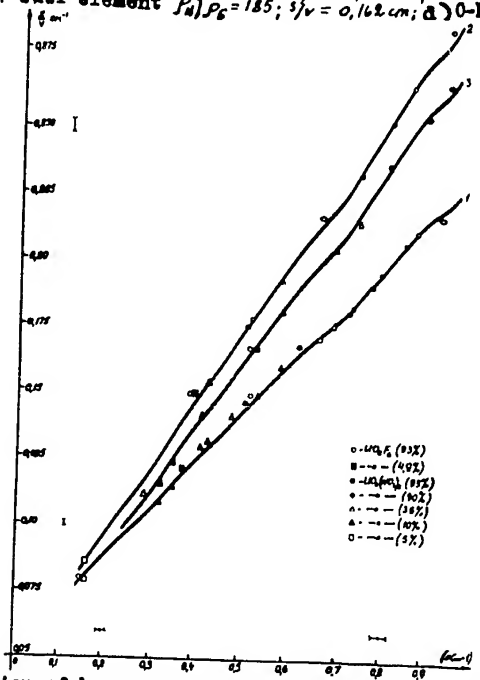

Fig.6 The systematization of homogeneous cylindrical critical assemblies.
1) - bare critical assemblies. 2) - critical assemblies with full H_2O reflector.
3) - critical assemblies with three-sided H_2O reflector.

359

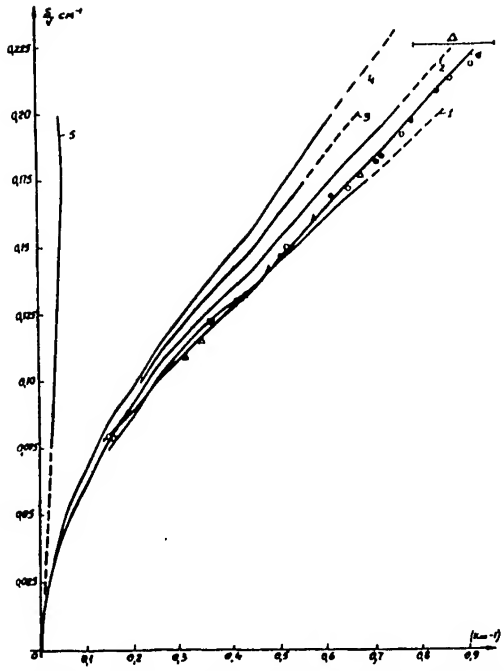

Fig.7. The comparison of calculational and experimental values of $\delta/\nu = f(K_{\infty} - 1)$
1,2–ρ– calculation using (13) at λ_0=2 and 3cm, respectively.
3,4–ρ– calculation using (14) at λ_0=2 and 3cm, respectively.
5 – contribution of thermal neutron leakage.
6 – results of systematization of bare cylinders.

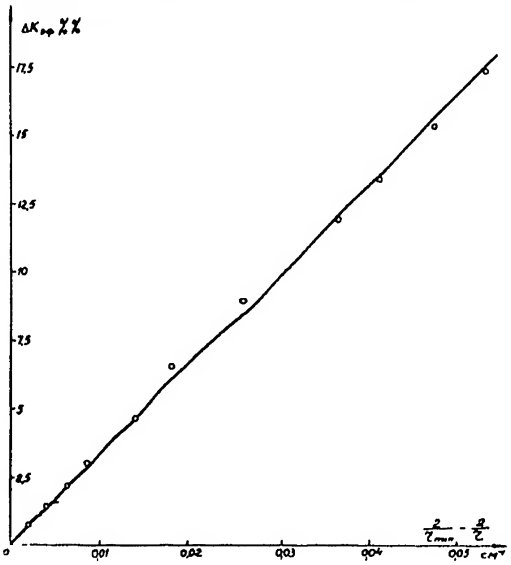

Fig.8. The dependence of reactivity excess on the ratio change of the sides core surface to its volume. K – 17 fuel element; $\rho_H / \rho_S = 1.85$